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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Report Nos. 50-272/84-23
50-311/84-23

Docket Nos. 50-272
50-311

License Nos. DPR-70
DPR-75

Licensee: Public Service Electric and Gas Company
80 Park Plaza
Newark, New Jersey 07101

Facility Name: Salem Nuclear Generating Station - Units 1 and2

Inspection At: Hancocks Bridge, New Jersey

Inspection Conducted: June 9, 1984 - July 6, 1984

Inspectors: for J. C. Linville, Senior Resident Inspector — 7.23.84 date

for R. J. Summers, Resident Reactor Inspector — 7.23.84 date

for E. H. Gray, Reactor Engineer, M&PS — 7/23/84

Approved By: L. J. Norrholm, Chief, Reactor Projects Section No. 2B, Projects Branch No. 2, DPRP — 7/23/84

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Inspection Summary
Inspections on June 9, 1984 - July 6, 1984 (Combined Report Numbers 50-272/84-23 and 50-311/84-23)

Areas Inspected: Routine inspections of plant operations including: status of previous inspection items, review of periodic and special reports, licensee event report review, operational safety verification, surveillance observations, maintenance observations, operating events, refueling activities, and allegation followup. The inspection involved 171 inspector hours by the resident NRC inspectors and 7 hours by one region based materials piping engineer.

Results: One violation involving failure to establish mechanical snubbers surveillance testing procedures was identified (paragraph 6). In addition, several other concerns were discussed including the resolution of the air gap problem associated with No. 23 Containment Fan Coil Unit and the safety classification of the work orders for installing No. 13 and No. 23 feedwater flow nozzles (paragraph 2), the plans to repair the crack in the suction line for No. 2 unit charging pumps and to prevent similar occurrences by aggressively pursuing timely corrective action (paragraph 8), the administrative controls associated with the placement of fuel in No. 1 reactor vessel and the deletion of the use of the Dillon load cell during replacement of Unit 1 reactor vessel internals (paragraph 9), and the resolution of allegations concerning security guard pat down searches and health physics technician practices (paragraph 10).

DETAILS

1. Persons Contacted

Within this report period, interviews and discussions were conducted with members of licensee management and staff as necessary to support inspection activity.

2. Status of Previous Inspection Items

(Open) Inspector Followup Item (50-311/84-15-04) This item involved the evaluation of a failure of No. 23 Containment Fan Coil Unit (CFCU) to start on April 17, 1984 apparently due to broken stator strap welds. The motor was replaced and the CFCU operated satisfactorily until June when it failed to start in slow speed. The motor was replaced with the motor from No. 14 CFCU after the end bells were reversed to place the junction box on the proper side for the Unit 2 configuration. This motor also failed to start in slow speed. The licensee then swapped the motor from 22 CFCU to 23 CFCU and it operated properly. This motor was selected because it was in original factory condition. The 3 others used in 23 CFCU since January 1984 had either the end bells reversed since they had been moved from Unit 1 or rotors which were new or had been rewound due to a previously identified problem of rotor bar cracking documented in paragraph 9 of Inspection Report 50-272/84-04. All three motors had evidence of stator-rotor rubbing. At a Station Operations Review Committee (SORC) meeting, which the inspector attended on June 12, 1984, the licensee concluded that the failures of 23 CFCU had probably been caused by stator to rotor rubbing due to improper air gap which cannot be measured due to the motor design. SORC decided that periodic low speed starts from a stopped condition are necessary to demonstrate CFCU operability for three months until longer term evaluation and corrective action can be completed. The long term program will involve increasing the tight 0.038 inch air gap tolerance, providing inspection ports in the end bells to permit measurement of the air gap and a test program to be developed by Franklin Institute to define the failure mechanism and to provide a capability to predict impending failures using vibration data. The licensee also tested the remaining Unit 2 CFCUs in low speed from a dead stop. The inspector will review the SORC minutes and the results of the evaluation and corrective action during a subsequent inspection.

(Open) Violation (311/84-15-05) The licensee has not yet responded to this violation involving misclassification of safety related work orders on feedwater system components following the April 6, 1984 water hammer event. One work order involved the replacement of the damaged No. 23 feedwater flow nozzle with the No. 13 feedwater flow nozzle since Unit 1 was in the midst of a refueling outage. At the time the inspectors understood that the welds connecting the section of piping containing the feedwater flow nozzle would be treated as non-safety related while the sensing lines connecting the feedwater flow nozzle to the reactor protection system would be reclassified as safety related. While reviewing the work order for replacement of No. 13 feedwater flow nozzle the inspectors found that this was the case only from the isolation valves to which the tubing connects, not from the feedwater pipe attachments to the isolation valves. The inspectors pointed out that this does not appear to be consistent with the FSAR statement that all equipment from the sensors to the trip breakers or initiation circuits of Engineered Safety Features are part of the safety related Reactor Trip System. The licensee maintained that the piping including the nozzle connections had not originally been bought or installed to safety-related requirements and that the replacement piping should therefore not be treated as such. In addition, IEEE 279-1971 to which the reactor protection system is designed indicates that the flow transmitter is the sensor according to the licensee. Finally, the licensee indicated that verification of proper installation of the flow nozzle will be accomplished during calibration of the feedwater flow channel. The inspector will review the calibration procedure during a subsequent inspection.

3. Review of Periodic and Special Reports

Upon receipt, the inspectors reviewed periodic and special reports. The review included the following: inclusion of information required by the NRC; test results and/or supporting information consistent with design predictions and performance specifications; planned corrective action for resolution of problems, and reportability and validity of report information. The following periodic reports were reviewed:

-- Unit 1 Monthly Operating Report for May 1984

-- Unit 2 Monthly Operating Report for May 1984

-- Unit 2 Cycle 2 Startup Test Report

-- Units 1 and 2 1984 Annual Environmental Operating Report (Non-Radiological)

4. Licensee Event Report (LER) Review

The inspectors reviewed LER's to verify that the details of the events were clearly reported. The inspectors determined that reporting requirements had been met, the report was adequate to assess the event, the cause appeared accurate and was supported by details, corrective actions appeared appropriate to correct the cause, the form was complete and generic applicability to other plants was not in question.

Unit 1

*84-11 No. 2 Fire Suppression Pump Inoperable For Greater Than Seven Days

*84-12 Charging/Safety Injection Throttling Valves - Disks Becoming Detached From Stems

*84-13 Loss of All 4KV Group and Vital Busses - Units 1 and 2

Unit 2

*84-13 Reactor Trip from 100% Due to Personnel Error While Testing

*84-14 Containment Ventilation Isolation - Inoperable

*84-15 Reactor Trip From 10% During Unit Shutdown Operations

*Denotes onsite followup

Unit 1

84-11 This report details the inoperability of the No. 2 Fire Suppression Pump for a period in excess of 7 days. The pump was taken out of service for required testing and maintenance; however, upon its return to service, the pump discharge valve failed closed which resulted in exceeding the 7 day action statement requirement. The valve failed due to separation of the disk from the stem as a result of a broken stem. The valve was replaced in kind. The valve type is a 12 inch gate valve from the Smith Valve Corporation (Mark No. K-77, Type 3620). Due to a prior history of similar failures on this type of valve, the licensee will conduct an engineering review to determine the failure mechanism. The inspector will review the supplemental report and additional corrective actions during a future inspection (272/84-23-01).

84-12 This report details failures of three of 24 total throttle valves in the Safety Injection system on both Units 1 and 2. This type of valve, a 1 1/2 inch Rockwell International globe valve, has failed similarly as reported in Licensee Event Reports 50-272/84-010-00 and 50-311/84-001-00. Prior failures have been on the reactor coolant system RTD loop manifold isolation valves. These failures resulted in additional investigation which revealed the problem on these Safety Injection throttle valves. Additional details of these events are documented in NRC Inspection Reports 50-272/84-04, 50-311/84-04 paragraph 9a; 50-272/84-08, 50-311/84-08 paragraph 4; 50-272/84-15, 50-311/84-15 paragraph 4; and, 50-272/84-19, 50-311/84-19 paragraph 9b. The inspector will review replacement of the remaining throttle valves during a subsequent inspection (272/84-23-02 and 311/84-23-01).

84-13 This report details a loss of off-site power on June 2, 1984 due to an operator error during an operation to isolate a portion of the switchyard. Additional details of this event are discussed in NRC Inspection Report 50-272/84-19; 50-311/84-19 paragraph 9.

Unit 2

84-13 This report detailed a reactor trip on May 11, 1984 caused by the failure of an instrument and control technician to lift the leads for the control signal to the No. 22 steam generator water level control system while troubleshooting a problem with the control room level recorder. Consequently, when he input a high level signal, the feedwater regulating valve closed and the reactor tripped on low water level in No. 22 steam generator. Initial inspector review of this event is documented in paragraph 9 of Inspection Report 50-311/83-19. The inspector will review the corrective actions stated in the LER including improved troubleshooting procedures and training program coverage of this event during a subsequent inspection (311/84-23-02).

84-14 This report described the blocking of the containment isolation function on high containment gaseous activity during a containment pressure relief in violation of Technical Specifications. Inspector review of this violation is detailed in Inspection Report 50-311/84-22. Inspector followup of licensee corrective action is being tracked under open item (311/84-22-01).

84-15 This report detailed a reactor trip from about 10 percent power while shutting the unit down due to failure of the high intermediate range flux circuitry to reset before power was reduced below the P-10 setpoint. Initial inspector review of this event is documented in paragraph 9 of Inspection Report 50-311/84-19. The inspector will review the licensee corrective action including an on the spot change to the operating procedure and adjustment of the reset value for the intermediate range high flux trip circuit during a subsequent inspection (311/84-23-03).

5. Operational Safety Verification

a. Control Room Observations

Daily, the inspectors verified selected plant parameters and equipment availability to ensure compliance with limiting conditions for operation of the plant Technical Specifications. Selected lit annunciators were discussed with control room operators to verify that the reasons for them were understood and corrective action, if required, was being taken. The inspectors observed shift turnovers biweekly to ensure proper control room and shift manning. The inspectors directly observed operations to ensure adherence to approved procedures.

b. Shift Logs and Operating Records

Selected shift logs and operating records were reviewed to obtain information on plant problems and operations, detect changes and trends in performance, detect possible conflicts with Technical Specifications or regulatory requirements, determine that records are being maintained and reviewed as required, and assess the effectiveness of the communications provided by the logs.

c. Plant Tours

During the inspection period, the inspectors made observations and conducted tours of the plant. During the plant tours, the inspectors conducted a visual inspection of selected piping between containment and the isolation valves for leakage or leakage paths. This included verification that manual valves were shut, capped and locked when required and that motor operated valves were not mechanically blocked. The inspectors also checked fire protection, housekeeping/cleanliness, radiation protection, and physical security conditions to ensure compliance with plant procedures and regulatory requirements.

d. Tagout Verification

The inspectors verified that selected safety-related tagging requests were proper by observing the position of breakers, switches and/or valves.

No violations were observed.

6. Surveillance Observations

The inspectors observed portions of the surveillance procedures listed below to verify that the test instrumentation was properly calibrated, approved procedures were used, the work was performed by qualified personnel. limiting conditions for operation were met, and the system was correctly restored following the testing:

-- SP(O)4.0.5 V test of the 21-24 MS 167's, Main Steam Stop Valves

-- 1PD16.2.011 Channel Functional Test of Source Range Channel 1N31

-- 2PD16.3.010 Channel Calibration Check of Power Range Channel 2N44

-- SP(O)4.7.15 Main Steam Isolation Valve Emergency Closure Time Response

-- Functional Testing of Mechanical Snubbers per Technical Specification 4.7.9.c

On July 3, 1984 the inspector reviewed the procedure being used to functionally test the mechanical snubbers for Unit 1. The first sample of snubbers were sent to Wyle Laboratory for testing. Based on the results, additional testing was required. On July 2, 1984, the Wyle mobile laboratory arrived on site to conduct the subsequent tests. The procedure being used was written by Wyle laboratory. No review or approval of this procedure by SORC was accomplished prior to implementation. In addition, no formal procedure had been implemented that controlled the functional testing of the mechanical snubbers to assure that all of the criteria of the Technical Specification surveillance requirements had been met. In accordance with Technical Specification 6.8.1, written procedures shall be established, maintained and implemented for surveillance activities for safety related equipment. Failure to establish formal, approved procedures to control the functional testing of the mechanical snubbers is a violation of the Technical Specifications (272/84-23-03). During preparations to make the vendor laboratory operational, the inspector asked representatives of the Salem Station QA and the in-service testing groups about the review and approval requirements associated with the vendor's procedures. Both representatives stated that they were treating the work activity as though it was being accomplished at the vendor's offsite facility and therefore needed no formal review and approval by the station. This appeared not to be consistent with the licensee's review and approval policy on subcontractor's procedures which affect safety related equipment as stated in Administrative Procedure (AP) 3, Document Control Program. Subsequent to implementation of the vendor's snubber testing procedure, the licensee did perform the required SORC and QA review and management approval.

7. Maintenance Observations

a. The inspector(s) observed portions of various safety-related maintenance activities to determine that redundant components were operable, these activities did not violate the limiting conditions for operation, required administrative approvals and tagouts were obtained prior to initiating the work, approved procedures were used or the activity was within the "skills of the trade," appropriate radiological controls were properly implemented, ignition/fire prevention controls were properly implemented, and equipment was properly tested prior to returning it to service.

b. During this inspection period, the following activities were observed.:

-- Troubleshooting 23 CFCU high speed breaker trips per Work Order No. 0099023040

-- Troubleshooting Unit 2 Containment Pressure Protection Channel III per Work Order No. 0099021552

-- Replacement of relays in 1C Safeguards Equipment Control Cabinet per Work Order 954561 and DCR 1EC1816

-- Installation of No. 13 feedwater flow nozzle per Work Order 84-06-06-936-8

8. Operating Events

A. Unit 2

On July 3, 1984 at 2:45 a.m., during functional testing of the control rods, the control bank C, group 1 rods could not be moved back to the fully withdrawn position. An urgent bank failure alarm occurred as a result of an erratic fuse in the moveable gripper circuitry at 4:25 a.m. The licensee commenced a shutdown from 100% by borating. The licensee determined the cause of the urgent rod failure at 7:20 a.m. and terminated the shutdown at 7:35 a.m. with power at 86%.

At 8:22 a.m. on July 5, 1984, the licensee declared an unusual event and commenced a shutdown of the unit from 100% power because of through wall cracks in the 8 inch charging pump common suction line near vent valve 2 CV372. The unit was off-line at 9:42 a.m. and subcritical at 9:48 a.m. The unusual event was terminated at 3:30 p.m. when cold shutdown boron concentration had been achieved and the unit was in hot shutdown (less than 350 degrees F). Cold shutdown was reached at 3:38 a.m. on July 6, 1984. The resident inspector and the region based materials-piping engineer observed the pipe leak and reviewed preliminary plans for repair of the leak and proposed inspections by the licensee of similar configurations. The leak was in the 8 inch diameter schedule 10 charging pump suction line at the edge of the weld attaching vent valve 2CV-372, FA-27. The leak appeared to be a nearly straight narrow opening approximately 2 inches in length crossing the heat affected weld zone of the valve attachment and oriented parallel to the 8 inch pipe axis. Water was squirting from near both ends of the defect upward approximate three inches.

The repair plan is to replace a portion of the 8 inch pipe with material of increased wall thickness (schedule 40) and to shorten the vent line. Fourteen other similar attachments to charging pump suction lines and the line in the vicinity of these attachments are scheduled for liquid penetrant examination (PT). The value of this examination for identification of other potential leak sites presumes that this type defect originates from the pipe outside surface. Visual examination of the inside surface and outside surface of the known leak area is planned to assure that cracking did originate from the outside pipe surface. The intention is to provide the defect area for metallurgical examination following this preliminary visual examination. Repair of the pipe section and initial examination of similar attachments will be followed by licensee periodic inspection of attachment welds and pipe hangers. The inspector will review the licensee evaluation and corrective action further during a subsequent inspection (311/84-23-04).

In reviewing this event the inspector noted from the Senior Shift Supervisor's (SSS) log that a "pinhole" leak was initially reported and a priority B work order was written on the 4 to 12 shift on July 4, 1984. According to the SSS and the Unit 2 shift supervisor, the report made by an equipment operator indicated that the leak was in the vent line and was similar to other charging system vent and drain line leaks seen in the past on both units. However, subsequent discussions and a review of the equipment operator's turnover sheet indicated that he was aware the leak was in the 8 inch line, but that his verbal report had been misunderstood. Since the work order was classified priority B requiring action as soon as possible, but generally interpreted to mean within 48 hours, a maintenance supervisor inspected the leak late the next shift. Shortly thereafter, a senior maintenance supervisor inspected the leak at the beginning of day shift and reported it to licensee management at about 7:45 a.m. on July 5, 1984. Technical Specification action statement 3.0.3 was entered due to the inoperability of all charging pumps and a plant shutdown was initiated at 8:22 a.m. The failure of the 4 to 12 operating shift personnel to thoroughly evaluate the leak and enter the action statement on July 4, 1984 is unresolved pending further discussions between the licensee and regional management (311/84-23-05).

9. Refueling Activities

The inspectors observed the refueling operations conducted by a Westinghouse refueling crew. During the fuel handling, various Technical Specification requirements were verified such as containment integrity, boron concentration, and nuclear instrumentation. Administrative and radiological controls were proper. While reviewing the video tape used for core verification, it appeared that the fuel assembly identification numbers could not be clearly verified in some cases. In addition, the personnel conducting the verification (video taping) appeared to refer to the core map on some occasions to identify the assembly. In discussions with the QC inspector witnessing the verification, the inspector determined that he was a contractor inspector who had never witnessed this activity before. He said that the resolution on the screen used during the verification was better than that of the videotape. The Senior Reactor Operator supervising the activity indicated that he only became involved on occasions when the numbers were difficult to read. The inspector did not interview the third individual involved in the verification, a member of the Westinghouse refueling crew. Since the FSAR takes credit for the strict administrative controls to prevent the infrequent fault of inadvertently loading a fuel assembly into an improper position, the inspectors questioned the lack of involvement of experienced licensee personnel in the actual verification. The reactor engineer was satisfied that all fuel had been properly placed by review of the videotape and that the nuclear instrumentation would detect any error as indicated in the FSAR. However, he indicated that a reactor engineer would be directly involved in future verifications. The inspector will verify this during a subsequent refueling outage (311/84-23-06).

The inspector also noted that the QC verification of the movement of fuel from the spent fuel pool to the reactor was performed by observing the movement of tags on the status board in the control room. However, no one appeared to check to ensure that the member of the Westinghouse refueling crew operating the crane in the spent fuel pool was picking up the fuel bundle from the designated coordinates. The inspector pointed out that this might be another way of providing more assurance that fuel is properly placed.

During a review of the Senior Shift Supervisor's log for the 3:00 p.m. to 11:15 p.m. shift June 21, 1984 the inspector noted that the licensee had made an on-the-spot-change to the reactor vessel (RV) reassembly maintenance procedure, M8C, to permit reinstallation of the RV internals without using the Dillon load cell to monitor crane loading while lowering the internals into position above the fuel. The inspector recalled that the licensee had taken credit for its use during this lift in its May 11, 1984 submittal to NRR regarding Control of Heavy Loads. The head of the Westinghouse refueling crew recommended that the lift not be made without the Dillon Load Cell. But the licensee engineering staff provided a safety evaluation to justify the one time procedure change which indicated that the requirements of NUREG 0612 were met without the Dillon Load Cell, that there were numerous other steps taken to assure proper crane operation, that the cable would carefully be monitored for slack, and that the allowable inaccuracy in the Dillon Load Cell are such that it does not provide added damage prevention during replacement of the internals. In subsequent discussions with the licensee engineering staff, the inspector found that they were much more concerned about the use of the Dillon Load Cell to detect binding which is more likely to lead to excessive crane loading during removal of the RV head and internals. While the inspector did not consider this to be a prudent change, the RV internals were re-installed without incident without the Dillon Load Cell. The inspector also questioned the use of the on-the-spot procedure change in this case because it appears to change the intent of the procedure by removing a device which might provide early warning of a problem with the lift before a slack cable would be evident. In cases where the intent of a procedure is changed on-the-spot changes are not permissible because additional safety review by SORC is required prior to implementation. The licensee committed to a careful review of their criteria for and method of implementing on-the-spot changes at an enforcement conference documented in Inspection Report 50-311/84-25.

10. Allegation Followup

On June 11, 1984, a contractor employee alleged that ten security guards had set off the metal detector at the auxiliary access point while coming on shift on June 6, 1984 and had not been patted down as required by security procedures. At 5:50 a.m. on June 22, 1984 the inspector witnessed the passage of the oncoming guard shift through the metal detector. About 50 percent set off the alarm, but all that did were properly patted down. This allegation was unsubstantiated.

On June 28, 1984, a contractor health physics technician alleged that the licensee intended to destroy a Loss of Radiological Controls (LRC) report that he had written on June 26, 1984 on an incident during which he had observed a licensee health physics technician improperly entering the Source Locker Room without dosimetry and without signing the appropriate radiation exposure permit. As of the end of the inspection period LRC 84-116 involving the alleged incident had not been completely dispositioned but was contained in the file. The inspector will review the final disposition of LRC 84-116 during a subsequent inspection (272/84-23-04).

11. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, violations or deviations. The unresolved item identified during this inspection is discussed in paragraph 8.

12. Exit Interview

At periodic intervals during the course of this inspection, meetings were held with senior facility management to discuss inspection scope and findings. On July 6, 1984, the inspectors met with licensee representatives and summarized the scope and findings of the inspection as they are described in this report.